ANL-7543

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LOCATING AND IDENTIFYING THE SOURCE OF THE MAY 24, 1967 FISSION-PRODUCT RELEASE IN EBR-II

by

R. R. Smith, D. W. Cissel,
C. B. Doe, E. R. Ebersole,
and F. S. Kirn

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Printed in the United States of America
Available from
Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U. S. Department of Commerce
Springfield, Virginia 22151
Price: Printed Copy $3.00; Microfiche $0.65

ANL-7543
Reactor Technology

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Argonne, Illinois 60439

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EBR-II Project

April 1969

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ABSTRACT

The first verified release of fission products to the primary sodium coolant in EBR-II occurred on May 24, 1967. Primary annunciation was given by the FGM (fission gas monitor), which indicated an approximately 3000-fold increase in the concentration of short-lived krypton and xenon fission-product species in the argon cover-gas system. Confirmatory evidence of the release was provided by analyses conducted for ^{133}Xe and ^{135}Xe in the cover gas, and for ^{131}I and ^{137}Cs in the primary coolant.

By removing groups of suspect subassemblies, followed by operation of the reactor, the origin of the fission-product release was eventually attributed to an experimental UO_2-PuO_2 fuel element (HOV-4) in irradiation subassembly XO11.

I. INTRODUCTION

Until May 24, 1967, approximately 30,000 driver elements and approximately 200 encapsulated experimental fuel elements had been irradiated in EBR-II without any detectable release of fission products to the primary system. Prior to the fission-product release on May 24, considerable effort had been devoted to studies of the annunciation, location, and implications of fuel failure, but in the absence of any practical experience with failures of fuel-bearing capsules or elements most of the effort had been oriented strongly in theoretical and conjectural directions.

The implications of experiencing an actual, verified failure for the first time are both immediate and long-ranged. The failure gave increased assurance that fission products can be released without damaging the system. On a broader and more long-ranged basis, the failure reinforced the implication that the consequences of specific types of failure may be accepted without interrupting the operation of the reactor.

With the increased emphasis currently being placed on the irradiation of experimental fuel elements in EBR-II, it is anticipated that problems associated with fuel failure will occur with increasing frequency over the foreseeable future. Anticipating also that the knowledge gained during the May 24 fission-product release will be of general interest to other investigators in the fast-reactor program, it seems worthwhile to document, in comprehensive form, those events which had an immediate bearing on the actual release.

Insofar as possible, pertinent information in this report is presented in approximate chronological order to reconstruct the logic applied to locate the failure. For a better understanding of the problems associated with locating a fuel failure, brief descriptions are given of driver and encapsulated fuel elements. Also described are the various annunciation and detection techniques as well as the use of information resulting from these techniques in diagnostic efforts.

II. DESCRIPTION OF PERTINENT CORE FEATURES

Because a detailed description of the EBR-II plant is given in the original Hazards Summary Review[1] and its Addendum,[2] only those features which have an immediate bearing on fission-product releases will be described here.

A. Plant Features

The EBR-II consists of an assembly of subassemblies, each of which is a basically hexagonal, Type 304 stainless steel tube containing driver-fuel elements, encapsulated irradiation specimens, or depleted-uranium blanket elements. The arrangement of the core and blanket subassemblies with respect to other components of the primary system is illustrated in Fig. 1. The entire system is immersed in a tank containing approximately 86,000 gal of sodium. Two centrifugal pumps, submerged in the sodium, deliver coolant at 700°F to the inlet plenum at a combined rate of approximately 9,000 gpm. By throttle valves and a stepped lower grid plate, the inlet stream is divided into a high-pressure system, which cools the core region, and a low-pressure system, which cools the radial blanket.

Both systems discharge into a common upper plenum from which the heated coolant, whose temperature is approximately 832°F at 45 MWt, flows to an intermediate heat exchanger submerged in the primary sodium. The coolant leaves the heat exchanger at approximately 690°F (at 45 MWt) and discharges directly back to the bulk sodium.

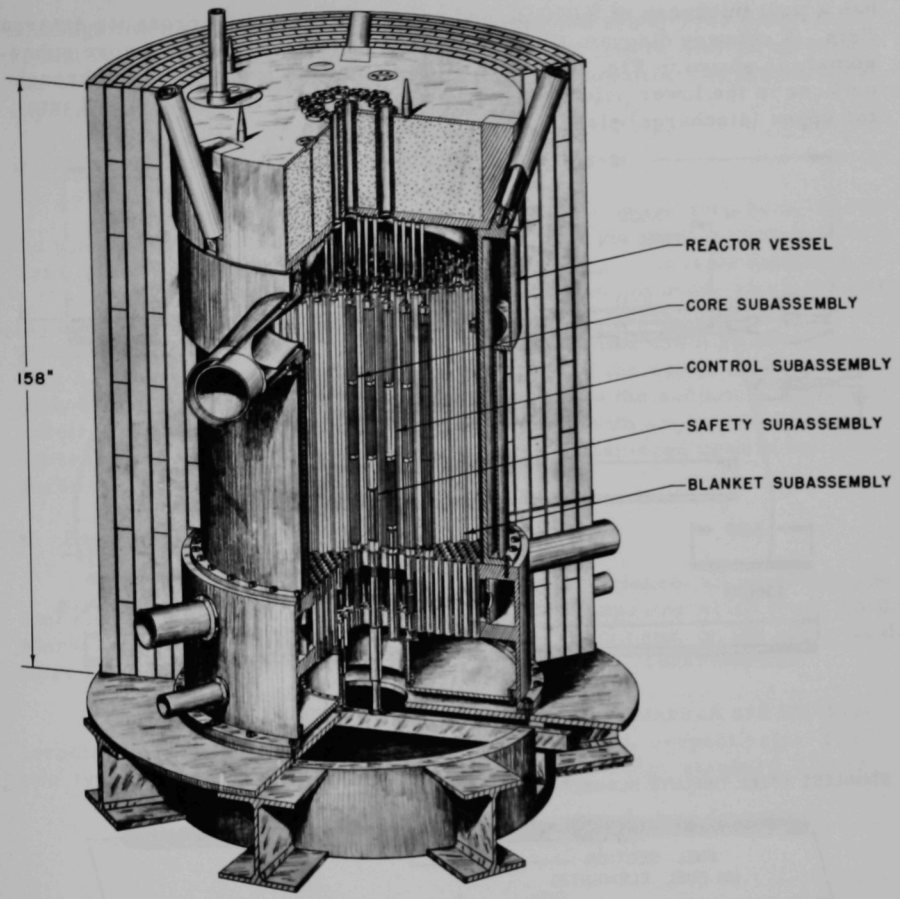

Fig. 1. Cutaway View of EBR-II

Above the upper surface of the sodium and beneath the upper structure is an argon-filled cover-gas plenum, approximately 14 in. high and 26 ft in diameter; its volume is approximately 625 ft^3. Under normal operating conditions, the cover gas contains minute quantities of rare-gas fission products generated from a small contamination of core components with fuel material.

B. Core Subassembly

The hexagonal tube of each core subassembly is filled with a close-packed hexagonal array of 91 standard Mark-IA fuel elements. The tube

has a wall thickness of 0.040 in. and measures 2.290 in. across its external flats. A cutaway diagram illustrating pertinent features of the core subassembly is shown in Fig. 2. Coolant flows from the lower plenum, through orifices in the lower pole piece, upward through the fuel bundle, and into the upper (discharge) plenum.

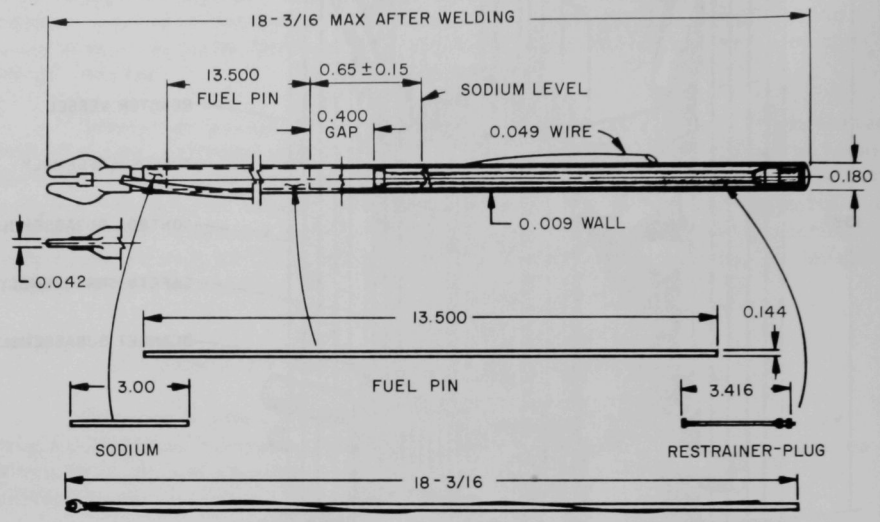

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Fig. 2. EBR-II Mark-IA Fuel Element and Subassembly (all dimensions in inches)

C. Control and Safety Rods

The hexagonal tube of each control and safety rod has a 0.040-in.-thick wall and measures 1.908 in. across external flats. Each contains a hexagonally pitched bundle of 61 standard Mark-IA driver elements. Both the control and the safety rods move vertically in hexagonal thimbles that have the same cross-sectional dimensions as a standard driver subassembly.

Although the maximum complement of control and safety rods consists of 12 and 2, respectively, one control rod at the time of the May 24 release had been replaced by a special stainless steel-filled drop rod for kinetics experiments.

D. Driver Elements

A standard driver element consists of a 13.50-in.-long by 0.144-in.-dia pin of U-5 Fs alloy enriched to 52.4 w/o ^{235}U. The pin is contained in a Type 304 stainless steel jacket, or cladding, having an outside diameter of 0.174 in. and an inside diameter of 0.156 in. A sodium bond, which occupies the annulus separating the fuel material and the cladding, serves as a heat-transfer medium. Details of a typical driver element are also shown in Fig. 2. Under room (STP) conditions, the level of the sodium bond is nominally 0.65 ± 0.15 in. above the fuel material. Above the sodium is a gas reservoir, approximately 0.663 ml in volume, filled with argon. As burnup proceeds and as the fuel swells, bond sodium is displaced upward and reduces the volume of the argon cover gas.

E. Experimental Irradiation Subassemblies

At the time of the May 24 fission-product release, all experimental fuel elements were encapsulated in Mark-A fuel capsules which were inside Mark-A irradiation subassemblies. Material specimens, on the other hand, were irradiated in Mark-B irradiation capsules and subassemblies.

Pertinent features and dimensions of typical Mark-A and Mark-B irradiation subassemblies are given in Figs. 3 and 4, respectively. The two types of subassemblies are indistinguishable from standard driver

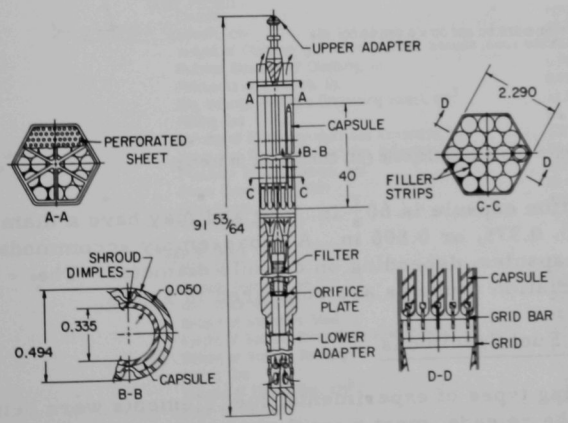

Fig. 3
Mark-A Irradiation Subassembly
(all dimensions in inches)

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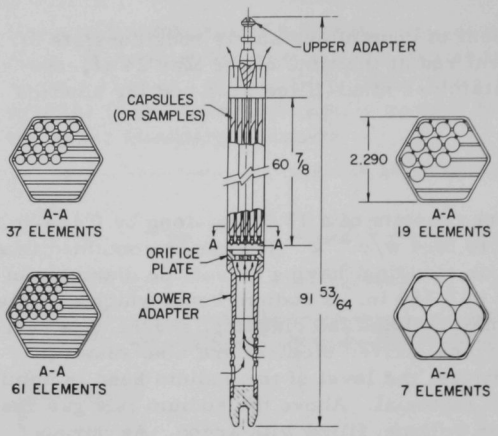

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Fig. 4. Mark-B Irradiation Subassembly (all dimensions in inches)

subassemblies in external features and dimensions. Internally, however, they are different from driver subassemblies and from each other. In the Mark-A subassembly, shroud tubes (illustrated in Fig. 3) surround the active length of the fuel capsule. In the Mark-B subassembly, on the other hand, 0.0925-in.-dia stainless steel wrapper wires, spot-welded to the capsules or specimens, serve as spacers.

F. Irradiation Capsules

A Mark-A irradiation capsule, illustrated in Fig. 5, consists of a Type 304 stainless steel tube, 0.020 in. thick, 0.375 in. in diameter, and 40 in. long. Each accepts a single experimental fuel element, which is sodium bonded to the capsule.

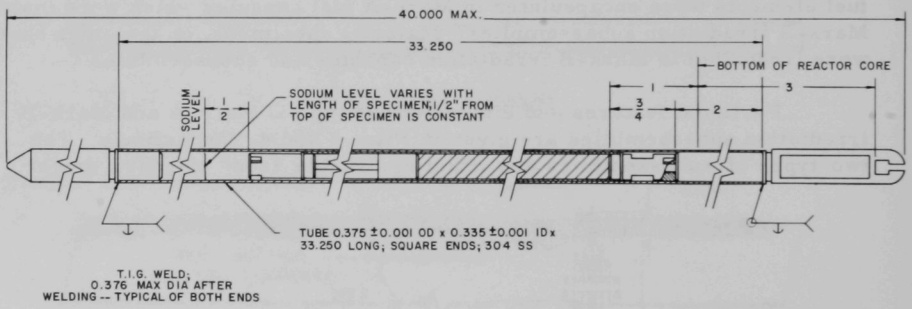

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Fig. 5. Mark-A Irradiation Capsule (all dimensions in inches)

A Mark-B irradiation capsule is $60\frac{7}{8}$ in. long and may have a diameter of 0.220, 0.250, 0.290, 0.375, or 0.806 in. A subassembly accommodates 61, 37, 19, or 7 of these capsules, depending on capsule diameter. The Mark-A and Mark-B irradiation capsules are compared in Fig. 6.

G. Typical Experimental Fuel Specimens

While many differing types of experimental fuel elements were being irradiated at the time of the release, most were fueled with mixed oxides of

uranium and plutonium. Pertinent information for a more or less typical oxide element is given in Table I for Capsule HOV-4, which was being irradiated in irradiation subassembly XO11 at the time.

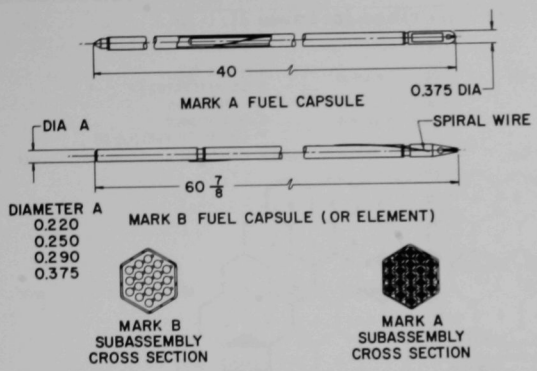

Fig. 6

Basic Designs of EBR-II Test Capsules Mark A and Mark B (all dimensions in inches)

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TABLE I. Physical Data for Experimental Capsule HOV-4

	UO_2-20 w/o PuO_2
Fuel Material	
Fabrication	Vibratory-compacted
^{235}U Content in Uranium, w/o	93.0
^{239}Pu Content in Plutonium, w/o	90.9
Length of Fuel Column, in.	11.5
Diameter of Fuel Column, in.	0.267
Weight of Fuel Material, g	92.2
Volume of Fuel Material, cm^3	10.54
Density of Fuel Material, g/cm^3	8.75 (80-85% of theoretical)
Weight of UO_2, g	73.8
Weight of PuO_2, g	18.4
Melting Point of Fuel Material, °C	2700
Calculated Center Fuel Temperature, °C	2850
Fuel Element	
Cladding	Hastelloy X
Length, cm	64.2
Weight of Cladding, g	58.0
Outside Diameter of Cladding, in.	0.295
Thickness of Cladding, in.	0.014
Gas Volume Available (including voids), cm^3	12.3
Filling Gas	He
Volume of Fission-product Gas Generated, cm^3	102.1
Volume of Fission-product Gas Released, cm^3	81.6
Pressure in Gas Plenum, psi	427
Power Generation, kW/ft	22.4
Hoop Stress, psi	4020
Thermal Stress, psi	14,500
Capsule (Mark A)	
Material	Type 304 SS
Outside Diameter, in.	0.375
Wall Thickness, in.	0.020
Weight of Stainless Steel, g	167
Height of Sodium Bond above Fuel Element, cm	3.8
Weight of Sodium Bond, g	9.0
Filling Gas	He and Ar
Volume of Filling Gas, cm^3	7.5
Hoop Stress, psi	460
Thermal Stress, psi	16,100

H. Core Loading for Run 25

The core loading at the time of the May 24 fission-product release, in Run 25, is illustrated in Fig. 7. The contents of all irradiation subassemblies in the core at that time are described in Table II.

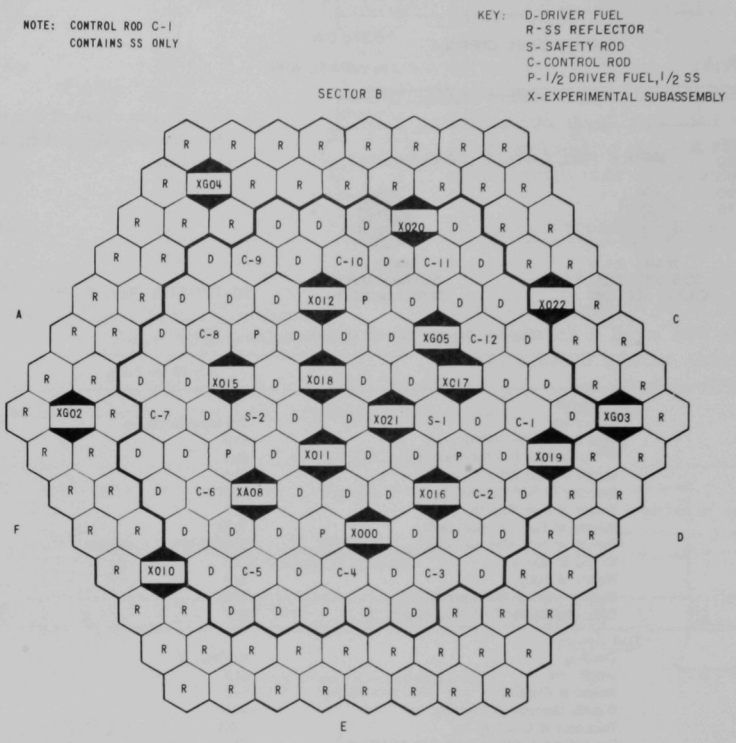

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Fig. 7. EBR-II Experimental Loading on April 17, 1967 (Run 25)

TABLE II. Experimental Subassemblies in Core on May 24, 1967

Subassembly Identification	Grid Location	Experimenter	Date Installed	Type of Fuel	Capsule Designation	Power Generation (kW/ft) Max	Power Generation (kW/ft) Min	Maximum Burnup as of 5/24/67 (a/o)	Cladding	Type of Subassembly
XG02	7A1	GE	7/16/65	UO_2-PuO_2	FOE	5.3	5.3	2.1	347 SS	Mark A19
XG03	7D1	GE	7/16/65	UO_2-PuO_2	FOA FOC	5.3	4.6	2.1	347 SS 347 SS	Mark A19
XG04	7B1	GE	7/16/65	UO_2-PuO_2	FOB FOD	5.3	4.6	2.1	347 SS 347 SS	Mark A19
XG05	4C2	GE	9/3/65	UO_2-PuO_2	F2C F2O* F2G F2V F2O F2R	15.5	13.5	5.9	Inconel 800 347 SS 347 SS 347 SS 347 SS 347 SS	Mark A19

TABLE II (Contd.)

Subassembly Identification	Grid Location	Experimenter	Date Installed	Type of Fuel	Capsule Designation	Power Generation (kW/ft) Max	Power Generation (kW/ft) Min	Maximum Burnup as of 5/24/67 (a/o)	Cladding	Type of Subassembly
XG05 (Contd.)	4C2	GE	9/3/65	UO_2-PuO_2		15.5	13.5	5.9		Mark A19
					F2T				347 SS	
					F2H				316 SS	
					F2X				316 SS	
		ANL		UC-PuC	HMV-5	19.3	18.8	5.5	304 SS	
					NMV-11				Nb-1 Zr	
					SMV-2				304 SS	
		ANL		U-Pu-Zr	NC-17	8.6	8.5	5.0	V-20 Ti	
					ND-24				V-20 Ti	
XA08	4F2	ANL	12/13/65	(Pu-U)C	HMV-1	26.0	17.2	4.9	Hastelloy X	Mark A19
					HMV-4				Hastelloy X	
					HWMP-1				Hastelloy X + W	
					HWMV-1				Hastelloy X	
					NMP-2				Nb-1 Zr	
					NMV-4				Nb-1 Zr	
					NMV-7				Nb-1 Zr	
					NMV-12				Nb-1 Zr	
					11 structural-materials capsules					
XO10	7F3	GE	3/24/66	UO_2-PuO_2	FOJ	8.6	7.7	2.3	Incoloy	Mark A19
					FOK				347 SS	
					FOL				316 SS	
					FOM				Incoloy	
		ANL			11 structural-materials capsules					
		PNL			4 structural-materials capsules					
XO11	2F1	ANL	5/9/66	UO_2-PuO_2	HOV-4	23.0	19.5	3.5	Hastelloy X	Mark A19
					HOV-10				Hastelloy X	
					HOV-15				Hastelloy X	
					SOV-1				304 SS	
					SOV-3				304 SS	
					SOV-7				304 SS	
					TVOV-1				V-20 Ti	
		GE		UO_2-PuO_2	F4A	17.9	16.4	3.5	304 SS	
					F4D				Inconel 800	
					F4E				316 SS	
					F4F				Inconel 800	
					F4G				316 SS	
					F4H				Inconel 800	
					F4J				316 SS	
					F4K				316 SS	
					F4L				Inconel 800	
		PNL		PuO_2-SS	5P9	11.5	7.5	4.1	304 SS	
					5P12				304 SS	
				UO_2-SS	5U14	5.9	5.9	3.3	304 SS	
XO12	4B2	NUMEC	8/10/66	UO_2-PuO_2	C1	15.5	13.5	2.1	316L SS	Mark A19
					C2				↓	
					C3					
					C4					
					D5					
					C6					
					C7					
					C8					
					C9					
					C10					
					C11					
					C12					
					C13					
					C14					
					C15					
					C16					
					C17				Hastelloy X	
					C18				Hastelloy X	
					C19				316L SS	
XO15	4A2	NUMEC	11/15/66	UO_2-PuO_2	B1	15.4	14.0	1.1	316L SS	Mark A19
					B2				↓	
					B3					
					B4					
					B5					
					B6					
					B7					
					B8					
					B9				Hastelloy X	
					B10				316L SS	
					B11				316L SS	
		GE		UO_2-PuO_2	F7C	14.0	14.0	1.0	316 SS	
					F7D				316 SS	
		ANL		(U-Pu)C	NMV-3	25.0	17.6	1.1	Nb-1 Zr	
					TVMV-1				V-20 Ti	
					HMV-2				Hastelloy X	
					NMP-1				Hastelloy X	
				Mark IA	BF02	7.6	7.6	0.6	Nb-1 Zr	
					BF03				Nb-1 Zr	

TABLE II (Contd.)

Subassembly Identification	Grid Location	Experimenter	Date Installed	Type of Fuel	Capsule Designation	Power Generation (kW/ft) Max	Power Generation (kW/ft) Min	Maximum Burnup as of 5/24/67 (a/o)	Cladding	Type of Subassembly
X016	4D3	GE ANL	11/15/66		10 structural-materials capsules 9 structural-materials capsules					Mark B19
X017	4C3	NUMEC	11/15/66	UO_2-PuO_2	A-1 thru A-11	15.4	13.5	1.1	316L SS	Mark A19
		UNC		(U-Pu)C	87	26.8	25.2	1.1	Incoloy	
					89				316 SS	
					90				Incoloy	
		ANL		Mark IA	BF04	8.4	8.2	0.6	304L SS	
					BF05				304L SS	
					BF06				304L SS	
					BF08				304L SS	
					BF09				304L SS	
					BF11				304L SS	
X018	2B1	GE ANL PNL	12/6/66		3 structural-materials capsules 3 structural-materials capsules 1 structural-materials capsule					Mark B7
X019	6D2	GE	1/13/67	UO_2-PuO_2	F8A	8.0	7.0	0.2	347 SS	Mark A9
					F8B				316 SS	
					F8C				304 SS	
					F8D				Incoloy	
					F8E				316 SS	
					F8F				316 SS	
					F8G				316 SS	
		UNC		(U-Pu)C	UNC-81	20.0	19.0	0.2	316 SS	
					UNC-82				316 SS	
					UNC-83				Incoloy	
		PNL			8 structural-materials capsules					
X020	6B5	GE	1/13/67	UO_2-PuO_2	F8I	8.0	7.0	0.2	316 SS	Mark A19
					F8J				304 SS	
					F8K				Incoloy 800	
					F8L				316 SS	
					F8M				316 SS	
					F8N				316 SS	
					F8O				347 SS	
					F8P				316 SS	
					F8Q				Incoloy 800	
		UNC		(U-Pu)C	UNC-84	20.0	19.0	0.2	316 SS	
					UNC-85				316 SS	
					UNC-86				Incoloy	
X021	2D1	PNL	2/25/67		7 structural-materials capsules					Mark B7
X022	7C4	PNL	2/26/67		7 structural-materials capsules					Mark B7

III. MONITORING SYSTEMS

At the time of the fission-product release, two on-line fission-product monitoring systems were operational. One, the FGM (fission gas monitor), senses the presence of relatively short-lived krypton and xenon fission products in the cover gas. The other, the FERD (fuel element rupture detector), senses the activities of delayed-neutron-emitting species in a bypass stream of primary coolant. Two other monitoring techniques, both off-line, were used at the time of the original release. In one, small samples of argon cover gas were periodically assayed for ^{133}Xe and ^{135}Xe. In the other, small samples of the primary coolant were radiochemically analyzed for ^{137}Cs and ^{131}I. Each of these methods is described briefly below.

A. Fission Gas Monitor (FGM)

Since a detailed description of the FGM is given elsewhere,[3] only those details relevant to an understanding of the system will be presented

here. The device consists essentially of the following components: a cover-gas delivery system, an electrostatic precipitation chamber, a water-filled trap, and an NaI gamma-pulse height analyzer.

Cover gas at a pressure of less than 5 psi and flowing at a rate of approximately 100 ml/min is pumped to the precipitation chamber, from which it is exhausted at near-atmospheric conditions to the reactor stack system. The precipitation chamber consists of two concentric cylinders: an inner one, 2 in. in diameter, of aluminum; and an outer one, 4 in. in diameter, of stainless steel. Running axially through the inner cylinder (the anode) is a 0.006-in.-dia stainless steel wire driven at a speed of approximately 18 in./min by a Bodine induction motor turning at 3 rpm. High voltage (1500 V positive) is supplied to the anode through a conventional Teflon-insulated high-voltage fitting.

To avoid the leakage of active cover gas to the surroundings from the inlet and outlet ports for the wire, a mercury seal in the form of a U-tube is installed on each port. The wire is pulled through both seals. A layer of water approximately 1 in. deep floats on the upstream leg of the seal on the outlet port for the wire.

The major cover-gas activities pumped through the precipitation chamber include ^{87}Kr, ^{88}Kr, ^{89}Kr, ^{133}Xe, ^{135}Xe, ^{138}Xe, and ^{41}A, all of which are beta active. At the instant of beta decay, each of the daughter species becomes a positively charged ion. Because of the electrical field, ionized species migrate to the traveling wire, where they are electronically neutralized since the wire is negatively charged with respect to the inner cylinder. The wire carries the electronically neutralized daughter species, now ^{87}Rb, ^{88}Rb, ^{89}Rb, ^{133}Cs, ^{135}Cs, ^{138}Cs, and ^{41}K, to the water layer in the downstream trap. All of the daughter species are alkali metals and react chemically with the water. The activity level of the water layer is monitored continuously with a conventional gamma-pulse height analyzer.

Of the seven alkali metal species fixed in the water layer, only three: ^{88}Rb, ^{89}Rb, and ^{138}Cs, contribute significantly to the gross activity level. The nuclides ^{41}K and ^{133}Cs are radioactively stable, and the half-lives of ^{135}Cs and ^{87}Rb are much too long (2.6×10^6 and 5.2×10^{10} yr, respectively) to allow any perceptible buildup. It follows that the activities sensed by the gamma-pulse height analyzer are exclusively those resulting from the decay of rare-gas fission-product daughters and that the discrimination against ^{41}Ar is essentially complete.

B. Fuel Element Rupture Detector (FERD)

In this system, a stream of molten sodium at a temperature of approximately 700°F is pumped at 100 gpm to the detection point through a

2-in. pipeline originating near the discharge side of the primary-secondary heat exchanger. Approximately 17 sec after leaving the core, the coolant flows through a 3-ft-long section of pipe, which is completely surrounded by graphite moderator bricks. Located in the graphite stack are various neutron-sensitive detectors, which monitor the neutron level in the coolant loop.

Three independent counting channels are used. Two are identical, and are each based on a single 2-in.-dia, 11-in.-long Reuter-Stokes BF_3 detector. The other channel consists of a gang of three 1-in.-dia BF_3 detectors and three 1-in.-dia boron-loaded chambers. The two identical channels are designated as Channels 1A and 2A; the third is identified as Channel B. For all channels, the signals are converted to count-rate form and are displayed as such on strip-chart recorders located in the control room.

C. Radiometric Analyses of Cover-gas Samples

In this method, three small samples (approximately 10 ml) of the primary argon cover gas are taken periodically (three per day) and analyzed under reproducible conditions of geometry with a multichannel gamma-pulse height analyzer. From measurements of the intensities of specific gamma peaks (81 keV for ^{133}Xe and 250 keV for ^{135}Xe), activity levels are established for the ^{133}Xe and ^{135}Xe fission-product species.

D. Radiochemical Analyses for ^{137}Cs and Iodine Fission Products

In this method, small samples (approximately 10 g) of the primary coolant are radiochemically analyzed for both ^{137}Cs and iodine fission products. At the time of the May 24 fission-product release, samples of primary sodium were taken from a point in the primary cold-trap loop. High radiation fields associated with ^{24}Na activity, both in the sampling area and in the samples, limit the frequency of sampling.

E. The Use of Fission-product Monitoring Techniques for Diagnosing Failures

The FERD system is sensitive to the presence of delayed-neutron emitters in the primary coolant and its response during fuel failure provides valuable information relevant to the type of failure. With a gas-type failure in either a driver element or in an experimental irradiation capsule, krypton and xenon fission products will be released to the coolant and will ultimately diffuse upward to the cover-gas plenum. Eventually the increased activity levels of rare-gas species will be detected by the FGM. Delayed-neutron-emitting species, presumably bromine and iodine, will remain chemically fixed in the sodium bond, which should remain undisturbed. The

FERD system, then, will fail to annunciate a gas-type failure. With a bond-type failure, on the other hand, both rare-gas fission products and delayed-neutron-emitting species may be released. Accordingly, both systems will respond.

Thus, if only the FGM system registers an increase in fission-product activity, it may be safely concluded that the failure involves the release of fission-product gases. If both systems respond, the failure may be identified as one in which sodium bond is lost.

In using the responses of the two systems for diagnostic purposes, care must be taken to consider the rate of bond release. If the rate of bond release is relatively fast (i.e., over a period of seconds or minutes), the considerations outlined above are valid. Such a release would be sensed by both the FERD and FGM systems. If, on the other hand, the release of bond sodium to the coolant occurs over a period of hours or days, the release rate becomes small compared with radioactive decay. Under these circumstances, the leak would not be annunciated by the FERD system and could even be "missed" by the FGM system. However, a very slow bond release would be sensed by an increase in the activity levels for ^{133}Xe and ^{135}Xe in the cover gas. Underlying this conclusion is the knowledge that relatively long-lived species such as ^{133}I and ^{135}I accumulate in the sodium bonds of normal driver elements and in the sodium bond between the damaged cladding of an experimental fuel element and its capsule. A slow bond loss in these cases releases ^{131}I, ^{133}I, and ^{135}I to the primary coolant. Even though ^{133}Xe and ^{135}Xe are not released in such a failure, the activity levels of these species in the cover gas will increase during and after a slow bond release because of their growth from their respective iodine parents. It follows that radiochemical analyses for iodine fission products in the primary coolant may also be used to indicate the existence of a bond leak that is too small to be annunciated by the FGM and FERD systems.

In principle and in practice, the four fission-product monitoring techniques provide sufficient information to permit important conclusions relating to the nature and the location of the failure. An illustration of the application of these techniques in the case of the May 24 fission-product release is given in Sect. V.

F. Testing and Calibrating the Monitoring Systems

Under routine operating conditions (i.e., with no defective fuel elements or irradiation capsules and with no fuel exposed to the coolant), all monitoring systems register outputs which reflect the generation of fission products from contamination of fuel material on the surfaces of fuel elements and from contamination of the primary coolant with natural uranium. With little exception, such levels are predictable and vary little from run

to run. In some respects, the existence of a constant fission-product background is desirable, if the background is not unreasonably high. In the first place, the ever-present and relatively constant background provides a satisfactory reference point against which the operational condition of the monitor(s) may be checked. In the second place, the background may be used as a reliable reference point in assessing the magnitude of an actual fission-product release.

To illustrate how ever-present background activity levels may be used as a convenient reference, it is instructive to consider the case of a given driver fuel element. Whenever the reactor is operated, fission products enter the sodium bond almost exclusively through recoil action. Because of the extremely short range of fission products in dense media, only a small fraction, approximately 0.05% of all fission products generated in the fuel material, enters the bond. Rare-gas fission products which enter the bond either through direct recoil or through their birth in the bond from their halogen parents diffuse upward and accumulate in the fuel-element plenum. A similar mechanism exists for gaseous fission products which are generated from tramp uranium in the system. In this case the rare-gas fission products diffuse upward into the cover-gas plenum where they accumulate.

Thus, activity levels for a given index species, say ^{88}Kr, in both fuel-element and cover-gas plenums follow buildup and saturation curves which have similar time dependencies. The amplitudes of the buildup curves for a specific index species in fuel-element and cover-gas plenums, however, depend in the former case on the power level and the range of fission products in the fuel material and, in the latter case, on the power level and the amount of tramp uranium. Thus, whenever the reactor is operated, a certain amount of ^{88}Kr is generated from tramp uranium sources and released to the cover-gas plenum while a different amount of ^{88}Kr is generated in the bond of a driver element and released to the fuel-element plenum. Whenever operation ceases or is interrupted, both components decay with the same half-life. As a consequence, the ratio between the ^{88}Kr inventories in the gas plenum of a given fuel element and in the cover-gas system is always a constant, regardless of time-power history. The same argument holds for any and all other gaseous index species, regardless of their half-lives.

The above mathematical facts provide a simple means for evaluating the maximum hypothetical signal to be expected from the release of all gaseous fission products from a given fuel element, regardless of its time-power history. From measurements of the FGM and FERD responses and the ^{133}Xe and ^{135}Xe activities in the cover gas before and after an unclad fuel pin is deliberately inserted in the core, it is possible to establish an upper limit for the signal-to-noise ratio, where the noise is defined as the normal index activity from tramp effects and the signal is defined as the

increase in signal caused by the recoil release of fission products from the exposed fuel. With the unclad pin in the core, a specific fraction of all the generated ^{88}Br and ^{88}Kr recoils directly into the flowing primary coolant, and the ^{88}Kr accumulates in the cover-gas plenum. In a clad pin, the same fraction of ^{88}Br and ^{88}Kr recoils into the bond, and the ^{88}Kr accumulates in the fuel-element plenum. On failure of the clad pin, the ^{88}Kr is released to the cover-gas plenum. Insofar as the responses of all fission-product monitors (which sense ^{88}Kr) are concerned, it is immaterial whether or not the ^{88}Br and ^{88}Kr activities are released continuously to the flowing coolant from the surface of an unclad pin or are the end product of ^{88}Kr accumulated in the fuel-element plenum and then released to the reactor cover-gas plenum by failure. The signal-to-noise (S/N) ratio at equilibrium will be the same in both cases, regardless of the time-power histories.

Two calibration tests, each using a single, unclad fuel pin in a special control rod, were carried out, one on May 21, 1965, and the other on May 11, 1966. As a consequence of these studies, it was shown that the maximum S/N ratio for a single fuel element located in a control-rod position in Row 5 was approximately 10. Such information proved valuable in eliminating failure of driver fuel as the origin of the fission-product release of May 24, 1967.

IV. CHRONOLOGY OF THE MAY 24 FISSION-PRODUCT RELEASE

A. Events prior to the Release

Up through and including Run 24, no known and verified fission-product releases had occurred in EBR-II. Following the Run-24 shutdown, two major changes were implemented in the reactor core and blanket.[4] First, the core was enlarged to accommodate a maximum of 91 subassemblies. Second, all depleted-uranium blanket subassemblies in Rows 7 and 8 were replaced by otherwise identical reflector subassemblies filled with stainless steel. On startup on April 21, 1967, it was noted that the power coefficient had decreased markedly. As a consequence, a great deal of effort was devoted throughout Run 25 to a variety of experiments designed to identify the origin of the power-coefficient decrease and to determine remedial action. The necessity for conducting such tests under a wide condition of power and flow resulted in an unusually complicated time-power history prior to the release. As an illustration, the time-power history from May 14 to the actual failure is summarized in Fig. 8. Finer details than those permitted in the illustration are summarized in Table III.

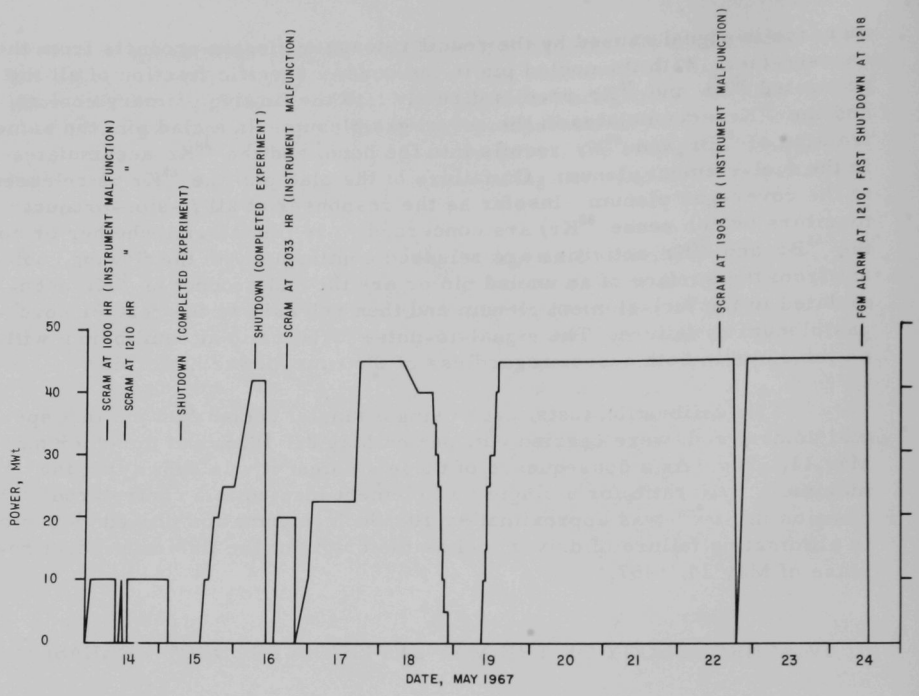

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Fig. 8. Time-Power History, May 14-24, 1967

TABLE III. Time-Power History, May 14-24, 1967

Date, 1967	Time	Power (MWt)	Date, 1967	Time	Power (MWt)
May 14	0000-0030	Increasing, 0-10	May 18	1140-1720	Level, 40
	0030-0930	Level, 10	(Contd.)	1720-1800	Level, 35
	0930-1330	Level, 0		1800-1900	Level, 30
	1330-2400	Level, 10		1900-1930	Level, 25
May 15	0000-0230	Level, 10		1930-2000	Level, 20
	0230-1345	Level, 0		2000-2030	Level, 15
	1345-1430	Increasing, 0-10		2030-2100	Level, 10
	1430-1545	Level, 10		2100-2230	Level, 5
	1545-1715	Increasing, 10-20		2230-2400	Level, 0
	1715-1800	Level, 20	May 19	0000-0900	Level, 0
	1800-2030	Increasing, 20-25		0900-1100	Level, 20
	2030-2400	Level, 25		1100-1130	Level, 25
May 16	0000-0450	Increasing, 25-42		1130-1300	Level, 30
	0450-0917	Level, 42		1300-2400	Level, 45
	0917-1300	Level, 0	May 20	0000-2400	Level, 45
	1300-1400	Increasing, 0-20	May 21	0000-2400	Level, 45
	1400-2000	Level, 20	May 22	0000-1900	Level, 45 (Scram)
	2000-2330	Increasing, 0-10		1900-2300	Increasing, 0-45
May 17	0000-0100	Increasing, 10-22.5		2300-2400	Level, 45
	0100-1450	Level, 22.5	May 23	0000-2400	Level, 45
	1450-1630	Increasing, 22-45	May 24	0000-1200	Level, 45
	1630-2400	Level, 45		1200	Level, 0
May 18	0000-0730	Level, 45			
	0730-1140	Decreasing, 45-40			

B. Fission-product Release of May 24

The five days prior to the May 24 release began with an incremental approach to the operating level of 45 MWt on May 19. Power of 45 MWt was reached at 1300 on that date and was held constant at this level until a scram at 1903 on May 22. During the following startup, power was increased over a period of approximately 4 hr, and reached a level of 45 MWt at 2300 on May 22. Operation continued at this level, without incident, until 1210 on May 24, when an alarm was received in the control room from the FGM. The alarm was set to respond when the signal from the FGM exceeded the normal equilibrium value (from tramp uranium) by a factor of three. Immediately following the alarm, the strip-chart recorders for each of the three FERD channels were inspected. No evidence of a FERD signal increase was noted. Meanwhile, the shift supervisor requested an additional sample of cover gas for analysis. When the results of a rough survey of the sample with a portable radiation-monitoring meter indicated a reading greater than ten times normal, a fast power setback to 50 kW was ordered at 1218. At 1226 an increase in air activity levels was noted at remote monitoring stations in the reactor building. At this time, 1226, the control rods were completely withdrawn, and the reactor building was evacuated by all personnel. At 1230 an alarm of a system monitoring activity in the air of the subbasement sounded. At 1245 personnel equipped with Scott Air Paks and appropriate protective clothing entered the reactor building to read instruments, collect air samples, and conduct a general area survey. The results of the survey are summarized in Table IV.

TABLE IV. Results of Radiation Survey of Reactor Building on May 24, 1967

Location	Normal Level	Level during Survey at 1245-1330 on May 24
Main-floor Air Monitor	50-150 counts/min	5000 counts/min
Main Floor, Average Level	<0.5 mR/hr	5 mR/hr
Depressed (Mezzanine) Area, max	2-3 mR/hr	250 mR/hr
Argon Vapor Trap	≈1.0 mR/hr	125 mR/hr
Effluent-discharge Monitor	1000-3000 counts/min	4000 counts/min
Air Sample, Main Floor	<0.1 mR/hr	35 mR/hr
Air Sample, Depressed (Mezzanine) Area	<1 mR/hr	60 mR/hr
Cover-gas Sample	<1 mR/hr	200 mR/hr
Spot Contamination--Shoe Covers, Gloves, etc.	<0.1 mR/hr	0.5-2 mR/hr

At 1250 the gamma monitor in the argon-purification cell alarmed at 20 mR/hr (normal reading is <1.0 mR/hr). Throughout the postrelease period, no measurable activity was indicated by the stack radiation monitor. Accordingly, ventilation of the reactor building continued unchanged.

Levels of radiation and air-particulate activity within the reactor building continued to rise until approximately 1620, when a definite trend toward lower values was noted. By 1115 on May 25, the results of analyses of air samples indicated airborne activities to be below maximum permissible concentrations for long-lived unidentified gamma-emitting isotopes. The counting of smear samples collected throughout the reactor building indicated a return to normal background levels. At this time, entry restrictions for the reactor building were lifted.

At no time prior to, during, or following the indicated time of release (established as 1150 on May 24) was there any evidence of an increase in FERD signal. Furthermore, no anomalous changes in reactivity, power, or subassembly outlet temperature were noted.

Confidence in the operational condition of the FERD system was given by the results of calibration measurements conducted three hours prior to and seven hours following the release. In both cases, all three channels responded in the usual test manner to the introduction of a calibrating neutron source.

C. Response of the Fission Gas Monitor

The release of fission products was so intense that essentially no information could be obtained from the FGM for approximately 40 min following the initial detection of the release. At approximately 1202, the FGM reading went off scale. Attempts to extend the readout capability of the system by changing scale settings on the scaler were futile. The largest scale change (a factor of 200) provided by the equipment, located in the reactor building, failed to bring the recorder pen back on scale. At 1226 the reactor building was evacuated, and efforts to bring the reading back on scale were abandoned. Eventually, personnel equipped with Scott Air Paks were permitted to re-enter the building and, in the course of conducting survey measurements, brought the FGM strip-chart readout back on scale by changing the gain of the amplifier. The results of an attempt to extrapolate the readings of the FGM that were made after the amplifier gain was changed and after subsequent scale changes are given in Fig. 9. A rough extrapolation of the reconstructed data to peak intensity resulted in a value of approximately 35,000 counts/sec. From a measured value of 12.5 counts/sec for the normal background level and the estimated value for the peak intensity, a S/N ratio of approximately 3000 was estimated. While no claims may be made for the reliability of estimated values, it is nevertheless clear that a sizable release had occurred.

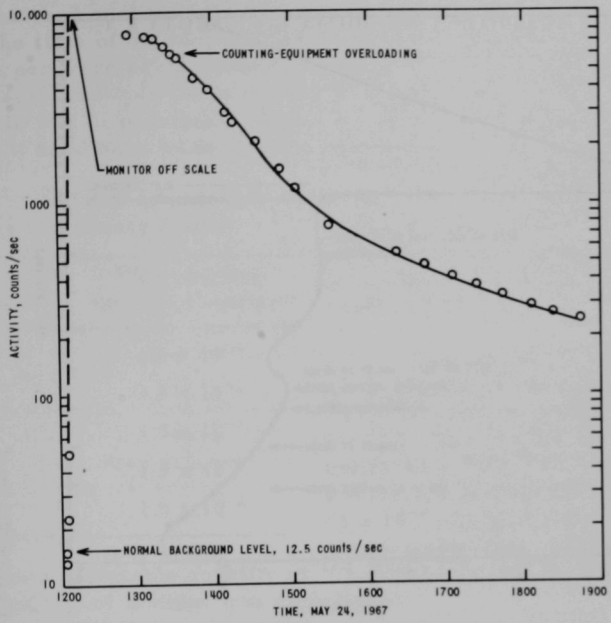

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Fig. 9. FGM Response on May 24, 1967

V. ANALYTICAL EFFORT

A. Radiometric Analyses for ^{133}Xe and ^{135}Xe in Cover Gas

Under normal operating conditions a 10-ml sample of cover gas is taken once every 8-hr shift and is analyzed with an NaI gamma-pulse-height analyzer. Intensity indices used for the ^{133}Xe and ^{135}Xe activities are based on the integration of count rates under the respective 83- and 250-keV photopeaks. The records of the ^{133}Xe and ^{135}Xe activities for the period May 15 through 28, 1967, are given in graphical form in Figs. 10 and 11, respectively. At the time of release, the reactor had been operating long enough (approximately five days) for the 9.2-hr ^{135}Xe activity to be close to its equilibrium value. The ^{133}Xe activity, having a half-life of 5.3 days, was still "growing in." From the data of Figs. 10 and 11, it may be seen that the S/N ratios for the ^{133}Xe and ^{135}Xe activities amounted to 100 and 400, respectively.

B. Radiometric Analyses for ^{137}Cs in the Primary Coolant

Under the operating conditions that existed at the time of the release, samples of primary coolant were taken approximately twice a month

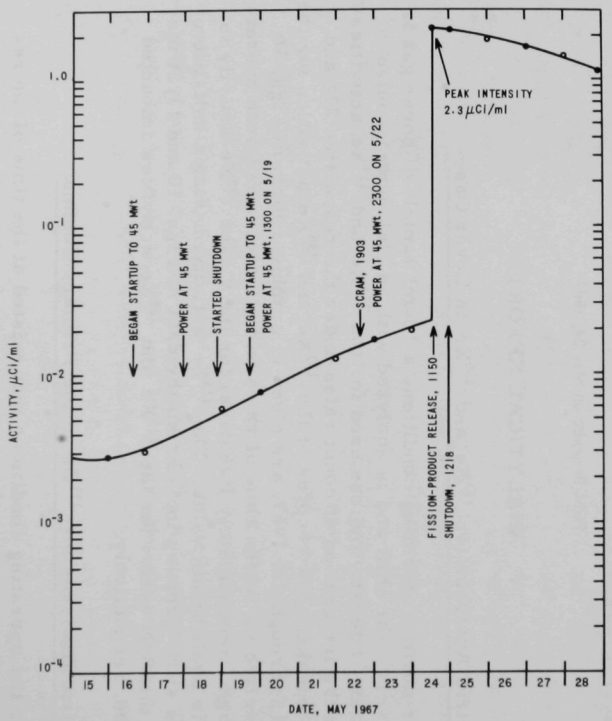

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Fig. 10. ^{133}Xe Activity, May 15 through 28, 1967

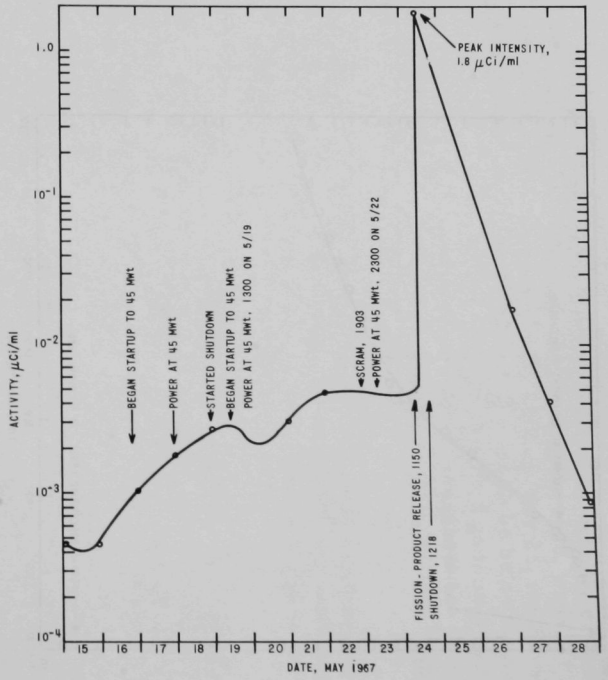

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Fig. 11. ^{135}Xe Activity, May 15 through 28, 1967

and were analyzed qualitatively with a multichannel NaI pulse-height analyzer. At the time of the May 24 fission-product release, intensive analyses for copper were being made of samples of primary coolant. On May 18, while analyzing a sample taken on May 8, it was noted that the ^{137}Cs content of the sample was unaccountably high. Because it was suspected that the sample might have been inadvertently contaminated with some other source of ^{137}Cs, a fresh sample was taken on May 19. The results of radiochemical analyses for ^{137}Cs of all primary-coolant samples that bear on the May 24 release are summarized in Table V.

TABLE V. ^{137}Cs Activities in Samples of Primary Coolant

Sample Date	^{137}Cs Activity ($\mu Ci/g$ of sodium)
4/24/67	$<5 \times 10^{-4}$
5/8/67	1.1×10^{-2}
5/19/67	1.5×10^{-2}
5/25/67	1.5×10^{-2}
6/1/67	1.6×10^{-2}

Radiochemical analyses were made of each of the samples for the $^{95}Zr/^{95}Nb$ fission-product pair and for ^{131}I. The results for all samples indicated a $^{95}Zr/^{95}Nb$ activity level too low for measurement (lower limit, $<5 \times 10^{-4}$ $\mu Ci/g$ of sodium). In only one sample--that collected on May 19--was a detectable quantity of ^{131}I noted. In this instance a value of 1×10^{-4} $\mu Ci/g$ of sodium was established.

C. Failure Diagnostics

Since the failure was the first definitely identified as such, essentially no precedent existed for diagnosing the nature of the failure and locating the subassembly containing the failed element (or capsule). From the start it was always clear that locating the failure would eventually require operation of the reactor after removing a group of suspect subassemblies. Failure to induce an additional release over an arbitrary period of time, for example 300 MWd, would indicate that the suspect was included in the group removed. Final identification would involve the reinsertion of the suspects, one at a time, until an additional release occurred.

In implementing the above procedure, careful consideration was given to the selection of a group of subassemblies which presumably would be small and which would include the most suspect subassemblies. The immediate problem, however, was to decide whether the failure was in a driver fuel element or an encapsulated experimental fuel element. Obviously, the elimination of driver fuel from the suspect list would narrow the field to the 12 irradiation experiments containing fuel (see Table II). As it turned out, sufficient information was available to conclude that driver fuel was not involved. In reconstructing the logic which led to this conclusion, it is instructive to summarize the information available:

1. the absence of a FERD signal;
2. the absence of anomalous changes in reactivity and power;
3. the absence of any indication of off-normal subassembly outlet temperature;
4. the 3000-fold increase in the FGM response;
5. the 100- and 400-fold increases in the signals for ^{133}Xe and ^{135}Xe, respectively;
6. the unusually large increase in the ^{137}Cs content of a sodium sample collected 16 days prior to the failure.

The absence of a detectable FERD signal prior to, during, and following the failure was accepted as evidence that (1) significant quantities of fuel material were not exposed, and (2) a rapid loss of sodium bond was not involved. From a knowledge of the S/N ratio established for a single unclad fuel pin from previous bare-pin tests[3] and assuming that a 20% increase in the FERD signal (above instrument noise) was significant and detectable, it was shown that any failure mode which caused the exposure of more than 2 cm^2 of fuel material would have been detected. (The area of a Mark-IA pin is 40 cm^2.) Exposed fuel areas of less than 2 cm^2 would very likely have escaped detection. The release of delayed-neutron-emitting species by a slow bond loss, however, would have been missed.

The complete absence of any anomalous change in either reactivity or power prior to, during, and after the failure was used as an indication that any effects of fuel melting and rearrangement that could have occurred as the result of failure were extremely minor. Similarly, the absence of anomalous indications in the exit temperatures of fuel subassemblies was interpreted as evidence that fuel melting and flow blockage were not involved, at least in any of the 19 fueled subassemblies located under thermocouples.

As indicated in Sect. IVC, the release caused an approximate 3000-fold increase in the FGM response. From the results of the previous calibration studies[3] with a single unclad fuel pin in a control rod, the maximum S/N ratio expected from the failure of a single element in Row 5 was known to be approximately 10. Thus, to explain the 3000-fold increase in terms of driver fuel failure, it was necessary to consider a mechanism involving at least 300 driver elements. In view of the complete lack of evidence suggesting either fuel melting, fuel exposure, or fuel-rearrangement effects, a multiple failure of this magnitude was not considered credible.

Similar conclusions were reached from the results of the ^{137}Cs analyses cited in Sect. VB. From the values of the specific activity of the samples collected after the failure (1.6×10^{-2} μCi/g of sodium) and the inventory of the primary sodium system (86,000 gal), the total activity of ^{137}Cs in the entire inventory was calculated to be 1.6×10^{11} dis/sec. The total amount of ^{137}Cs in the fuel material of a driver element irradiated to a burnup of 1.0 a/o corresponds to a disintegration rate of 7.1×10^{10} dis/sec (1.9 Ci). However, only a small fraction of the total fission-product inventory of a metallic fuel pin is released from the fuel material. By far the larger fraction remains with the fuel material and is not lost to the primary coolant.

The release fraction for the ^{137}Cs generated in a Mark-IA fuel pin was estimated first by measuring the ^{137}Cs activity in the bond sodium of a Mark-IA fuel element that had been irradiated to a burnup of 1.0 a/o. By using the result of this measurement and the total ^{137}Cs inventory in the fuel pin (obtained from burnup and fission-yield data), a value of 0.05% was estimated for the release fraction of ^{137}Cs from the fuel material to the sodium bond.

Thus, of the total of 7.1×10^{10} dis/sec of ^{137}Cs in a driver element at a burnup of 1.0 a/o, only 3.6×10^{7} dis/sec could have come from a single driver element. Thus, to explain the 1.6×10^{11} dis/sec of ^{137}Cs activity in the primary coolant inventory, it was necessary to postulate the complete loss of bond from at least 220 driver elements that had been irradiated to a burnup of 1.0 a/o.

Since all other evidence contradicted such a conclusion, attention was immediately focused on the possible failure of an encapsulated oxide-type fuel element which, because of higher operating temperatures and higher porosity, would release a much larger fraction of its volatile fission-product inventory. In fact, the fraction of the total rare-gas fission-product inventory released to the plenum of a highly irradiated UO_2-PuO_2 fuel element could easily exceed 50%.[5] It is easily shown that the 50% release of the rare-gas fission-product inventory from a single mixed-oxide fuel element (such as those irradiated in XG05 or XO11) could account for the increase in the ^{137}Cs content of the primary coolant.

VI. LOCATION AND IDENTIFICATION OF THE FAILURE

The inability to reconcile the magnitude of the release with a credible failure in driver fuel, coupled with the ability to explain the release in terms of a failure in a single encapsulated oxide-fuel element, essentially eliminated driver fuel from consideration. A review of all 16 experimental irradiation subassemblies was conducted. Of the 16, four were eliminated

because they contained only nonfueled irradiation specimens. Of the remaining 12, subassemblies XO11, XG05, and XA08 were considered the most suspect, primarily because they had been irradiated to the highest burnup.

A second criterion applied was that of cladding material; it was believed that fuels clad with so-called "exotic materials" were more likely to fail than those clad with conventional materials.

A. Fission-product Release of June 11, 1967

Meanwhile, a cautious attempt was made to restart the reactor and to reinitiate a gas release. On May 27, interlock checks were completed. All control and safety rods were first manually driven through their normal operating range to detect any evidence of binding. None was noted.

The scram times of all control and safety rods were measured in accordance with standard operating procedures. These times varied from +12 to -20 ms when compared with drop times measured at the start of Run 25 on April 21, 1967. The maximum drop time of 302 ms was well below the maximum allowable time of 325 ms. The drop time for the two safety rods varied from 594 to 900 ms, both values well within the allowable limit of one second.

All nuclear instrumentation and all area radiation-monitoring systems were checked and found to be satisfactory. Particular emphasis was placed on the FERD and FGM systems. Pu-Be and ^{137}Cs sources were used to test the respective responses. Both systems were shown to be responding satisfactorily.

A standard approach to criticality was made, and, from an extrapolation of subcritical counting data to criticality, it was shown that no significant change in core reactivity was indicated. Upon reaching criticality at 50 kW, a comparison was made between the measured reactivity balance and the balance which existed during a critical measurement taken on April 26. Although a net gain of 5 Ih was indicated, the change was not considered significant in view of a 70-Ih correction needed to compensate for fuel burnup in the intervening period.

As a consequence of the above subcritical and critical studies, it was concluded that no evidence of fuel-rearrangement effects was indicated and that no obvious damage had occurred within the core. Accordingly, on May 29, five days after the fission-product release, the reactor was restarted and brought to a power level of 2.5 MWt. After holding at this level for 13 hr with no indication of an additional release, the system was shut down.

The next step consisted of another attempt to reinitiate a fission-product gas release, this time at substantially higher levels of power. The reactor was restarted on June 4. Power was increased in 2.5-MWt increments, and held at each level for one hour and until a radiometric assay was completed for the ^{133}Xe and ^{135}Xe index species. Upon reaching 15 MWt on June 6, the power was held constant for 8 hr. When an analysis of all monitoring data revealed nothing unusual, power again was increased in 2.5-MWt increments. On June 9 a power level of 30 MWt was reached. Operations continued at this level until a scram at 1848 on June 9. An immediate restart was begun, and a power level of 30 MWt was again reached at 2335. At 2037 on June 10, a spurious signal from the FGM led to an anticipatory shutdown. After confirming from cover-gas analyses that the signal was spurious, the system was restarted and again brought to 30 MWt at 2246 on June 10. Operations continued at this level until 0223 on June 11, when the FGM registered a persistent increase in signal. Six minutes later, at 0229, the system was shut down. At this time, 115 MWd of power operation had been accumulated since the May 24 release.

A graphical summary of the FGM signal prior to, during, and following the June 11 release is given in Fig. 12. In this instance the release

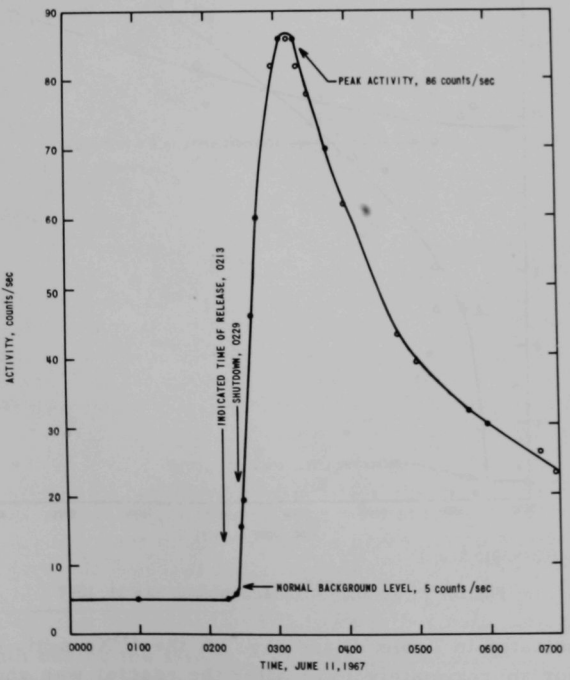

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Fig. 12. FGM Response on June 11, 1967

TABLE VI. ^{133}Xe and ^{135}Xe Activities[a] at Time of Fission-product Release of June 11, 1967

Time	^{133}Xe ($\mu Ci/ml \times 10^3$)	^{135}Xe ($\mu Ci/ml \times 10^3$)
0100	19.5	1.9
0216	Estimated Time of the Release	
0229	Reactor Shutdown	
0237	20.8	9.5
0325	20.2	10.0
0436	18.7	12.2
0900	18.6	18.1
1100	-	20.0
1650	22.6	25.6
1950	23.6	24.3

[a]Values corrected for decay to the indicated time of release.

was far less intense than that of May 24. From the information given in Fig. 10, it is easily shown that the S/N ratio amounted to only 16. At no time prior to, during, or following the indicated time of release was there any evidence of an increase in FERD signal or anomalous changes in reactivity, power, or subassembly outlet temperature.

The results of analyses conducted on cover-gas samples during the June 11 release are summarized in Table VI and illustrated in graphical form in Fig. 13.

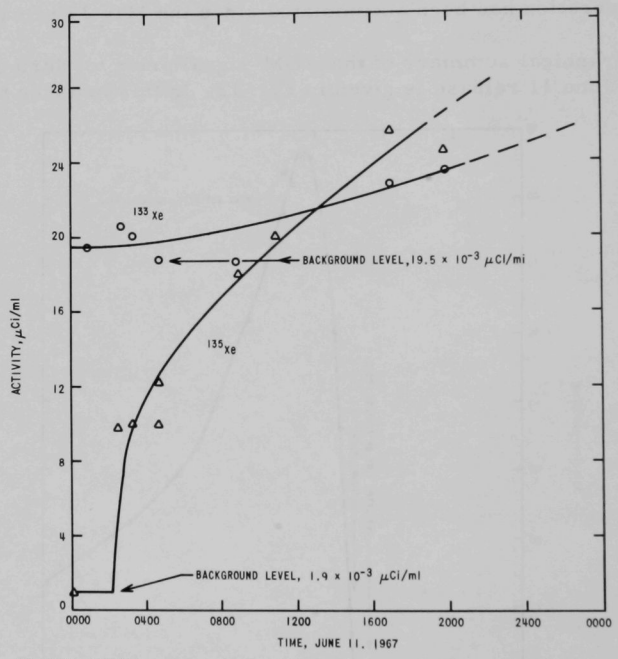

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Fig. 13. ^{133}Xe and ^{135}Xe Activities on June 11, 1967

As indicated in Table VI and Fig. 13, the ^{135}Xe activity continued to increase for approximately 15 hr after the reactor was shut down. Such behavior suggested that the failure involved the release of gaseous fission

products (as indicated by the sharp increase in the ^{135}Xe activity between 0100 and 0237) and the release of sodium bond that contained the parents of ^{135}Xe in the form of the chemically fixed ^{135}I. The apparent fractionation between the ^{133}Xe and ^{135}Xe activities, as judged by the respective S/N ratios, was attributed to: (1) a substantial voiding of the gas inventory of the suspect during the first release on May 24, (2) a significant holdover of the longer-lived ^{133}Xe in the cover-gas plenum, and (3) the almost-complete rebuildup of ^{135}Xe in the suspect immediately prior to the June 11 release. The fact that fresh fission products were released was also indicated by the FGM. As indicated above, the S/N ratio for this system was established as 16, a value reasonably consistent with a S/N ratio of 12 noted for the increase of ^{135}Xe activity.

B. Fission-product Release of June 19, 1967

Following the June 11 release, the activities of both ^{133}Xe and ^{135}Xe in the cover-gas system were reduced by purging the system with argon for approximately one week. On June 19 the reactor was restarted. Power was increased in 2.5-MWt increments to a power level of 10 MWt. At 2229 on June 19, after the reactor had accumulated only 16 MW-hr of operation since startup, the FGM signal began a gradual but persistent increase at a rate higher than that expected for normal buildup. By 2231, the signal had increased from a background level of 2.0 to approximately 23 counts/sec. At this time a shutdown was initiated. Special cover-gas samples were taken and analyzed; the results confirmed that another release had occurred. Data taken from the FGM strip-chart recorder at the time of the release are summarized in tabular form in Table VII and illustrated in graphical form in Fig. 14.

TABLE VII. FGM Data at Time of Fission-product Release of June 19, 1967

Time	FGM Reading (counts/sec)
2220 (6/19)	2 (background level)
2229	10
2231 (shutdown started)	23
2257	460
2315	580
0000 (6/20)	270
0100	180
0200	135
0300	100
0400	75
0500	62
0600	46

The increase in the FGM response after reactor shutdown is characteristic of the system. A sudden increase in the concentration of the rare-gas parents is not reflected in the FGM signal as a step increase because the activities of the daughter species must be integrated in the water leg of the FGM.

An illustration of the increases noted for the ^{133}Xe and ^{135}Xe activities in the cover gas is given in Fig. 15. The increases in the activities after shutdown reflect the release of bond sodium during the failure. The respective parents, ^{133}I and ^{135}I, are extruded with the bond and upon decay generate additional ^{133}Xe and ^{135}Xe.

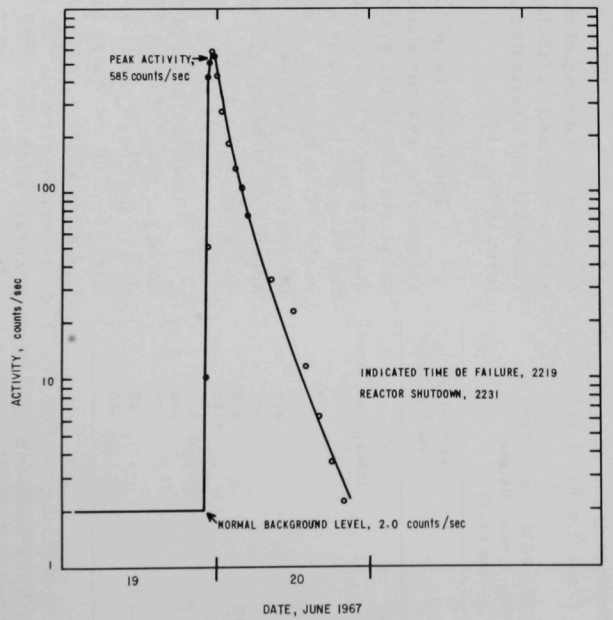

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Fig. 14. FGM Response on June 19 and 20, 1967

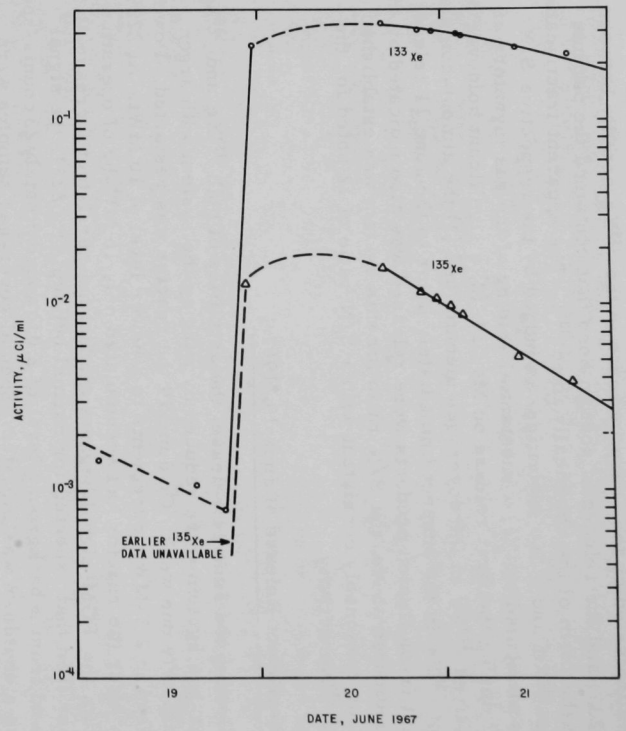

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Fig. 15. ^{133}Xe and ^{135}Xe Activities, June 19-21, 1967

The large S/N ratio observed for the FGM system and the ^{133}Xe activity, along with the relatively low level of power needed to initiate release (10 MWt), were interpreted as evidence that the defect was increasing in size.

As indicated above, three experimental subassemblies (XG05, XA08, and XO11) were considered prime suspects. Accordingly, all three were removed from the core. In the course of fuel handling, each subassembly was lifted and held in the raised position for several minutes on the fuel-handling gripper in an attempt to initiate an auxiliary release by reducing the static sodium pressure. After each lift and hold, gas samples were taken, and analyzed for ^{133}Xe and ^{135}Xe activities. In no case was an increase in activity noted.

C. Operating the Reactor without Suspect Subassemblies

Following the removal of the three suspect experimental subassemblies, the reactor was started up. Criticality was achieved at 2115 on June 21. Power was increased in 2.5-MWt increments, holding at each level for approximately one hour while gas samples were taken and analyzed. On June 22, power was increased to 30 MWt and held at this level for five days. At no time during this period was any evidence of a fission-product release noted. Interpreting this as evidence that the suspect had been removed, the reactor was shut down on June 27 in preparation for the next step, which called for the insertion of the suspects one at a time.

D. Fission-product Release of June 28, 1967

Following the shutdown of June 27, experimental subassembly XO11 was reinserted. On the following day, the reactor was started. Power was increased in 2.5-MWt increments, and upon reaching a power level of 7.5 MWt, the fourth fission-product release in the series was realized.

Graphical summaries of the response of the FGM system and the activity increases for ^{133}Xe and ^{135}Xe are given in Figs. 16 and 17, respectively. In this instance the activities appeared to be heavily weighted in favor of the shorter-lived species. A S/N ratio of 220 was noted for the FGM, and values of 5.0 and 1.2 were established for the ^{133}Xe and ^{135}Xe activities, respectively. The heavy fractionation in this case was attributed to the facts that the shorter-lived activities were near equilibrium levels at the time of release and that the reactor had not been operated long enough to generate significant quantities of longer-lived species.

E. Operating the Reactor without Suspect Subassembly XO11

Immediately after the release the system was shut down. Subassembly XO11 was removed, and subassemblies XG05 and XA08 were reinserted.

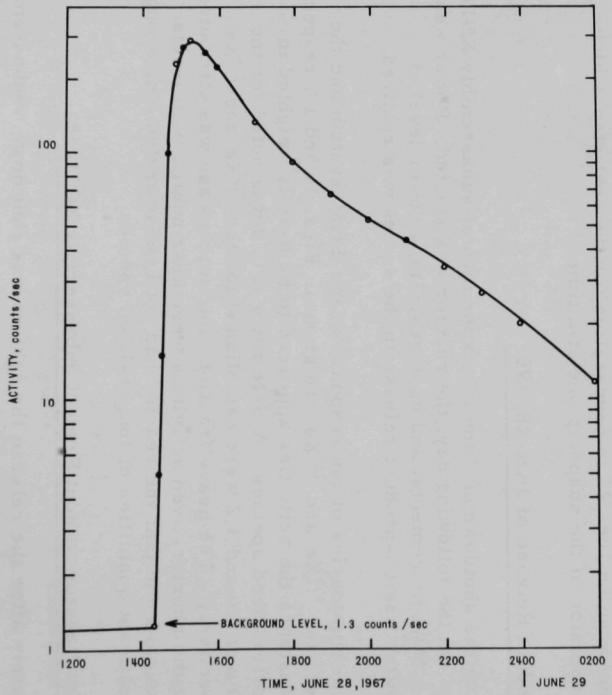

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Fig. 16. FGM Response on June 28, 1967

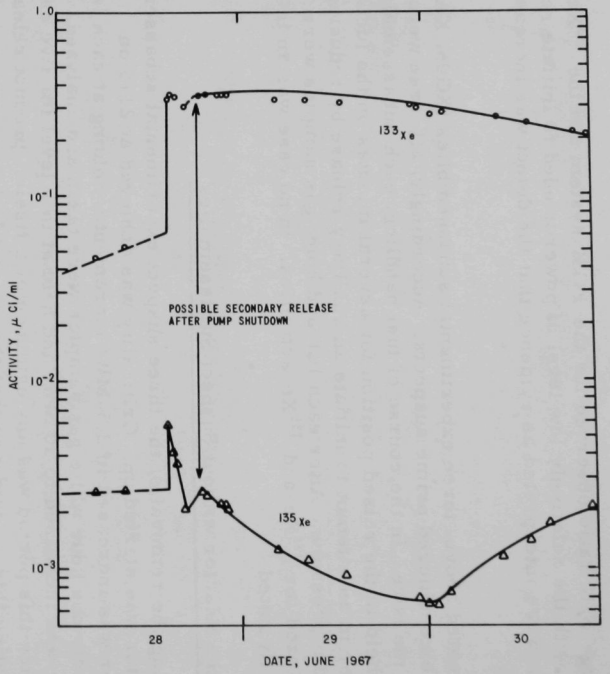

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Fig. 17. ^{133}Xe and ^{135}Xe Activities, June 28-30, 1967

The reactor was restarted on June 29. Power was increased in 2.5-MWt increments until a power level of 30 MWt was reached on June 30. Operations continued at this power level for approximately five days. No evidence of an additional fission-product release was indicated. The system was shut down on July 4, restarted on July 5, and brought to 45 MW on July 6. Prolonged operation of the system at 45-MWt without additional evidence of a fission-product release was regarded as proof that XO11 was the source of the releases.

F. Attempts to Induce an Additional Gas Release in the Interbuilding Coffin (IBC)

The fact that no fission-product release was noted with subassembly XO11 removed from the core, coupled with the knowledge that its reinsertion (followed by power operation) led immediately to an additional release, was interpreted as almost conclusive evidence that XO11 contained a defective fuel capsule. To provide additional evidence and assess the usefulness of out-of-pile identification techniques, consideration was given to using the IBC as a leak detector. Essentially, the technique was based on transferring XO11 to the IBC, partially exhausting the IBC argon atmosphere with a conventional roughing pump, and monitoring the IBC argon cooling gas for evidence of ^{133}Xe. Presumably, residual ^{133}Xe in the gas region of the capsule would expand through the defect and be detected in the radiometric analyses of the cooling gas. Accordingly, XO11 was transferred to the IBC, which was moved to the interbuilding passageway between the reactor building and the FCF (Fuel Cycle Facility). The tests involved reducing the internal pressure of the IBC in 3-in.-Hg increments between 25 (atmospheric pressure) and 5 in. After each pressure reduction, 15 min were allowed to elapse before gas sampling. In no case was any evidence of gas release noted.

From the readings of thermocouples located at the exit side of the subassembly and in the gas-recirculation loop, it was apparent that bond sodium had frozen and was apparently sealing the defect. In an attempt to melt the sodium, insulated heating wire with a gross thermal output of 1200 W was installed around the inlet recirculation pipe. While inlet gas temperatures rose eventually to 255°F, the outlet gas temperature never exceeded 140°F (melting point of sodium, 208°F). Nevertheless, tests were conducted at pressures as low as 5 in. (Hg). In no case was gas release noted. Accordingly, it was surmised that the bond sodium which had frozen and plugged the defect had not remelted.

In a second series of tests conducted in the reactor building, additional attempts were made to maintain the bond temperature above the freezing point. For 15 min prior to transferring subassembly XO11 from the fuel-unloading machine (FUM) to the IBC, hot argon gas (200-350°F) from the FUM was circulated through the IBC in an effort to preheat its

inner skirt. Shortly after transferring XO11 to the IBC from the FUM, the pressure in the IBC was reduced rapidly to 2 in. Hg, the system was backfilled immediately with tank argon, and gas samples were taken from the IBC. No evidence of a gas release was noted. Although the bond sodium was molten upon entry into the IBC (as indicated by thermocouple readings), the 15 min elapsing between entry and final vacuum pulldown might have been too long to prevent cooling and subsequent freezing of the sodium.

G. Subsequent Disposition of Subassembly XO11

Following the second series of IBC tests, subassembly XO11 was returned on July 21, 1967, to the storage basket in the primary reactor tank. On August 14 it was reloaded into the IBC, which was placed in the equipment air lock. Moist argon at a temperature of 70-80°F was circulated at a rate of 0.4 ft^3/min through the subassembly. After one hour the flow rate was increased to 7 ft^3/min. The subassembly was then dried by circulating air at a temperature of 70-80°F through it at a rate of 30 ft^3/min. After one hour of drying, the subassembly was transferred to the air cell of the FCF for disassembly, which consisted essentially of cutting away the hex tube and removing the irradiation capsules from the grid.

By August 16, all capsules had been removed. Each was wet-swabbed with a sponge to remove residual surface sodium and sodium oxide, and then was inspected for visual evidence of failure. Inspection operations were completed on August 21; no visual defects had been found.

Each capsule was subsequently neutron-radiographed. Of the 19 capsules, three (F4F, HOV-4 and HOV-10) were shown to contain damaged fuel elements. Each was characterized by relatively gross cladding faults, with the fuel and cladding swelled out to the inner capsule surface at many points. Dye-penetration tests were performed to extend the sensitivity of visual inspection methods. The results of these tests were, however, ambiguous.

Capsules HOV-4 and HOV-10 were returned to the Argonne-Illinois facilities for destructive examination, and capsule F4F was returned to its sponsor, General Electric Co., Sunnyvale, California. The results of examinations conducted on HOV-4 and HOV-10 revealed that, of the two, HOV-4 was very likely the source of fission products noted during May and June. Evidence indicating that the HOV-4 capsule had failed includes the following:

1. The oxide element had suffered gross damage.

2. The capsule released a small but significant amount of radioactive gas when placed in an evacuated container.

3. The capsule had decreased in weight during the irradiation by an amount approximately equal to the weight of the original sodium bond.

4. The rare-gas fission-product inventory of the capsule was less than 1% of that expected from calculations based on fuel burnup.

Although the exact location of the defect in HOV-4 was never unambiguously located, the fact that essentially all bond sodium had been extruded implies that the defect very likely was in the region of the lower weld.

The failure of capsule HOV-4 in irradiation subassembly XO11 on May 24, 1967, was the first verified failure of any fuel-bearing element or capsule in EBR-II. Inadvertently, the failure increased the assurances: (1) that fission-product releases can be annunciated promptly and monitored easily, and (2) that the reactor may be operated safely over a sustained period of weeks with a defective experimental capsule in it.

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